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MIRROR FUSION-FISSION HYBRIDS

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Abstract

The fusion-fission concept and the mirror fusion-fission hybrid program are outlined. Magnetic mirror fusion drivers and blankets for hybrid reactors are discussed. Results of system analyses are presented and a reference design is described.

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SUMMARY:

The fusion-fission hybrid concept offers the potential of being a versatile method of making the vast energy potential of U-238 and Th-232 available to man, and doing so at a reasonable cost.

At Lawrence Livermore Laboratory we have been studying hybrids based on magnetic mirror plasma confinement since the early 70's. This work started by studying the potential performance of hybrid blankets and has evolved to the point where we are now conceptually designing complete fusion-fission systems. These systems consist of mirror fusion components producing 14 MeV neutrons, fertile blankets in which the D-T neutrons provide energy and net fissile material, and fission burner reactors that convert this fissile material to energy. These studies indicate one hybrid reactor can supply makeup fissile fuel for as many as 10 fission burner reactors of equal thermal rating and that the capital cost ($/kWe) of such systems (fusion hybrid breeder + fission burner reactors) is only about 25% above the capital cost of thermal fission reactors alone.

Fusion hybrids could be a relatively near-term application of fusion research. Plasma experiments presently under construction, the Mirror Fusion Test Facility (MFTF), and the Tandem Mirror Experiment (TMX) scheduled for operation in 1981 and 1979, respectively, should come close to demonstrating plasma confinement needed for hybrid reactors.

In addition to expanding our fission energy resources manyfold, hybrids could offer the flexibility needed to make alternate fission fuel cycles practical.

The prospects for the fusion-fission hybrid are exciting and there is an expanding effort worldwide to quantify the hybrid's potential.

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I. INTRODUCTION:

What is Fusion-Fission?

Fuse a deuterium (D) and a tritium (T) nuclei and a 14 MeV neutron plus a 3.5 MeV alpha are produced. Fusion-fission is a concept for utilizing such fusion neutrons to cause fertile materials to fission directly and/or indirectly. Direct fission of fertile material is possible because the kinetic energy of D-T fusion neutrons is high enough to excite ^{238}U and ^{232}Th to fission. Indirect fission occurs because high energy neutrons can generate additional neutrons via (n,fiss), (n,2n), (n,3n) and $^{7}Li(n,n'T)$ reactions. These excess neutrons, in turn, can convert fertile into fissile material. Excess neutrons are needed because for each D-T fusion neuton produced a triton (T) must be bred, all or mostly by $^{6}Li(n,T)$ reactions.

All fusion-fission systems consist of fusion components producing, containing and maintaining a fusion plasma and a blanket that surrounds the plasma intercepting fusion neutrons. If net fissile material is produced in the blanket, the fusion-fission system may include separate fission reactors fueled with this fissile material. (Fig. 1)

While other neutron producing fusion reactors are possible; namely, D-D, the much higher cross section for D-T fusion at relatively low temperature makes D-T fusion the practical choice for fusion-fission.

Fusion-fission can be classified in a number of ways. One way is whether fission in the blanket is encouraged or suppressed. If fission in the blanket is encouraged, we classify it a "Hybrid." [1] If blanket fission is suppressed it is classified a "Symbiosis." [2] The principle nuclear reactions occurring in symbiotic and hybrid systems are outlined in Fig. 2. With the hybrid we can fission the fissile material produced in separate fission reactors and/or in the blanket. We think the best use of the hybrid is production of excess fissile material for use in separate fission reactors.

Why Fusion-Fission?

Fusion as we presently conceive it is neutron rich but energy poor. Fission, on the other hand, is energy rich but neutron poor. We, therefore, think fusion and fission can complement each other to the benefit of both.

Fusion-fission benefits fusion because fission will increase energy release per fusion event which will improve power balance and/or power density. This in turn reduces fusion technology requirements (lower magnetic *fields, lower beam energies, lower component efficiencies) needed for* economically competitive power generation. This will allow earlier commercialization of fusion and thus, an earlier return on the investment in fusion research.

Fusion-fission could benefit fission in a number of important ways. Fusion-fission could eliminate fission's dependence on ^{235}U by converting the world's abundant fertile resources (^{238}U and or ^{232}Th) to fissile fuel. This, in turn, could make present-day thermal fission reactor technology a long-term energy option. As an external source of neutrons fusion-fission could provide the fissile material needed to make alternate fuel *cycles practical. For example, ^{233}U producing fusion breeders* could be colocated with reprocessing plant(s) in secure international sites providing "denatured" fuel to national fission reactors. The "denatured" fuel could be ^{233}U plus ^{238}U or ^{232}U. Spent fuel would be returned to the international site for reprocessing. Any plutonium and other unwanted actinides produced in the fission reactors and/or in the fusion breeder could be consumed by recycling them through the fusion breeder. Fissile production can be independent of power generation and requires no initial fissile material inventory. Fusion-fission has no doubling time constraints, therefore, capacity can be built as rapidly as needed.

2. BLANKETS

A principal component of any fusion-fission system is the blanket that intercepts and utilizes fusion neutrons to produce tritium, fissile material and/or energy. The blanket's role can vary from just producing fissile material to a nearly critical fission assembly just producing energy. Examples of blanket types covering this range are displayed in Fig. 3. The first two are nonfission blankets with anticipated net performances covering the range expected for blankets of symbiotic systems.

Net performance means the space and time averaged number of fissile atoms and energy removed from the blanket per fusion neutron. Energy production is listed as blanket energy multiplication (M) which is the ratio of energy produced in the blanket to fusion neutron kinetic energy (14.1 MeV). Tritium is also produced to fuel the D-T fusion reaction. The first[2] is dependent [illegible] ^{7}Li (n,n'T) reactions for excess neutrons while the second[3] uses Be for neutron multiplication. A less expensive neutron multiplier such as Pb could be used but performance would be less.

The rest of the blankets listed in Fig. 3 use fission for neutron and energy multiplication and are classified as hybrid blankets. All except the [illegible] hybrid blanket has uranium in the first zone to take advantage of ^{238}U (n,fiss), (n,2n) and (n,3n) reactions. The first two hybrid blankets are based on single fuel cycles (U or Th) and produce single fissile products (^{239}Pu or ^{233}U). The third hybrid blanket type combines the advantage of ^{238}U fast fission with the advantage of ^{233}U as a thermal reactor fuel. The fourth is similar to the third except that Pu produced in the U zone is recycled in that zone. The more Pu recycled the higher the ^{233}U breeding ratio and the blanket energy multiplication will be. The fifth hybrid blanket type is representative of fission blankets that produce energy, not fissile material. This blanket consists of a fast fission U zone producing neutrons that drive a thermal lattice fission zone.

Our emphasis, to date, has been on uranium hybrid blankets. We have also [illegible] at a thorium hybrid blanket.

The physical motivation for the hybrid is the fact that fast neutrons will cause the abundant fertile isotopes, uranium 238 and thorium 232, to fission with the resultant release of energy and neutrons. At 14 MeV, the kinetic energy of a neutron generated by deuterium-tritium fusion, the fission cross section is [illegible] barns for uranium 238 and 0.37 barns for thorium 232. (Fig. 4) The number of neutrons generated per fission vs incident neutron energy is shown in Fig. 5.

For 14 MeV neutron induced fission the number of neutrons generated in ^{238}U is [illegible] and in thorium 232 is [illegible]. This fission cross section and neutron release data only suggests the possibility of significant energy and neutron multiplication. Infinite medium calculations show the actual theoretical potential for energy neutron multiplication of 14 MeV neutrons in uranium and thorium. Table 1 shows results of infinite medium calculations for natural uranium, uranium 238, and thorium. Here breeding reactions refer to [illegible] reactions in uranium 238 and uranium 233, respectively.

The energy multiplication and fissile breeding ratios predicted by the infinite medium calculations are exciting, but how good are such calculations?

To partially answer this question we have compared calculated and experimental results of a natural uranium pile.[4] Table 2 shows results of this comparison. The pile was a 106 centimeter long 99 centimeter diameter cylinder of 85% dense natural uranium with a 14 MeV neutron source at its center. The calculated fission and uranium 238 (n,γ) reactions both are within 7% of the measured experimental values. Based on this comparison we have some confidence in our ability to calculate hybrid blanket performance.

A thick pile of fertile material is not a hybrid blanket. There are numerous design considerations that must be addressed in order to develop a consistent blanket design. The important design considerations are: nuclear requirements, blanket geometry, refueling and replacement, tritium handling, heat removal, structural integrity, and materials. This list outlines topics that must be considered, all of which interrelate and affect nuclear performance.

There are two major nuclear requirements, tritium breeding and subcriticality. We believe the blanket should breed tritium sufficient to fuel the D-T fusion reaction. We also believe that the blanket should be subcritical under all conditions both normal and abnormal. The blanket geometry must conform to plasma and magnets and allow for penetrations. A uniform current flux of D-T neutrons into the blanket is also very desirable. The blanket geometry must also allow for blanket refueling and replacement. Tritium removal and containment methods are important considerations because choices made effect tritium breeding rate needed as well as type and quantities of permeation barriers used in the blanket. Heat removal and structural integrity both effect the amount of structure needed. Blanket performance is quite sensitive to the ratio of structure to fuel because of competition for the neutrons.

To get an idea of how blanket performance is affected by blanket requirements and design tradeoffs, we compare a blanket's performance to that of an infinite medium.

In Table 3 we see that requiring a uranium blanket to breed tritium and contains structure results in a plutonium breeding ratio of 2.2 compared to [illegible] for the infinite medium and energy generation of 200 MeV vs 309 MeV. Both total breeding (tritium plus fissile) and energy generation are reduced one-third compared to the infinite uranium medium.

Uranium metal has swelling problems at burnups and temperatures that are too low to be of much interest for hybrids. As shown in Table 4 the effects of using ceramic forms of uranium fuel are significant. Uranium dioxides performance is half that of uranium metal and uranium monocarbide is two-thirds of uranium metal. Since we are emphasizing fissile breeding we do not need the high burnup and temperature capabilities of ceramic type fuels. We, therefore, are concentrating on metallic types of uranium fuels such as uranium 7 weight percent moly and uranium silicide. Performance with these fuels are approximately 25% below the uranium metal case.

Blankets consist of fuel contained in a pressure vessel of some form. Our conceptual hybrid blankets consist of many individual pressure vessels containing a thorium or uranium fuel followed by tritium breeding lithium fuel. One such blanket module is shown in Fig. 6. A major feature of this module is a cylindrical pressure vessel capped by a hemispherical dome. To keep the thickness of this pressure vessel as thin as practical the diameter of the modules are relatively small and the pressure vessels are cooled by the inlet coolant, in this case helium, at approximately 300°C. After cooling the hemispherical first wall, the coolant reverses direction and flows through the interior of the module removing the nuclear heat generated in the fuel. Minimizing pressure vessel thickness, especially the first wall, is extremely important because the fusion neutrons must pass through this material before reaching the fertile fuel. A pressure vessel thickness of .5 centimeters (stainless steel) appears to be a reasonable compromise between nuclear performance and mechanical requirements.

Fission zone thickness is dictated by the required neutron leakage into the tritium breeding zone. Fission zone thickness varies betw en 20 and 30 centimeters for a blanket module of this type. Total blanket thickness is approximately one meter.

Plasma containment dictates blanket geometry. The blanket must not intrude into the plasma volume and must allow for fueling and exhaust penetrations. It is economically advantageous for the blanket to cover as much of the plasma as possible and to have a uniform D-T neutron current over its surface. For mirror confinement of the type portrayed in Fig. 7, a majority of the D-T neutrons are generated in a central spherical volume. To meet the objectives of high coverage and uniform current, the blanket is a spherical annulus surrounding the central plasma volume and is inside the coils. The plasma is surrounded by radially aligned cylindrical modules, except for where penetrations are provided for fueling and plasma leakage fans.

As depicted in Fig. 8 the blanket forms a spherical annulus that surrounds the plasma and that has penetrations for fueling and leakage. The degree of blanket coverage used is dependent on thickness of the blanket coolant plena and shield, as well as plasma and coil geometries. Blanket performance is strongly affected by coverage.

Fig. 9a shows the dependence of performance on coverage. Dropping coverage from 90 to 80 percent reduces fissile breeding and energy multiplication by approximately 40 percent. Performance is not linerally proportioned to coverage because tritium breeding must be maintained by increasing local tritium breeding. Fig. 9b demonstrates how this is done by reducing the thickness of the fission zone.

An extremely important aspect of blanket design is accounting for exposure effects. To illustrate this point Fig. 11 shows an example of the effects of exposure on a uranium monocarbide blanket. Exposure is measured in terms of integrated wall loading or energy current of 14 MeV neutrons across the first wall. M (blanket energy multiplication) is initially 8 and peaks at 32 after an exposure of 20 megawatt years per square meter, a factor of four increase. The tritium breeding ratio (T) doubles after 20 megawatt years per square meter. Pu the net plutonium ratio drops to 0 after an exposure of 24 megawatt years per square meter. These effects occur because the concentration of fissile plutonium and fission products builds up in the uranium. After an exposure of 20 megawatt years per square meter plutonium 239 concentration is 9% and the burnup is 12%. As stated earlier we require the blanket to be subcritical (unable to support a fission chain reaction) under all conditions. As you can see by what happens to M (blanket energy multiplication) this blanket never goes critical. The fission neutron multiplication factor K achieves a maximum value of 0.7 at 20 MWy/M exposure. K at 0 exposure is 0.3. Criticality by reconfiguration must also be avoided and it is this that is expected to set an upper limit to exposure. Until reconfiguration scenarios are examined, we are setting an arbitrary upper limit on the fissile buildup of 4%. As it turns out, economics dictates the blanket be removed at fissile buildup of less than 3%.

The effects of fractional blanket coverage and exposure must be combined to determine effective blanket performance. Table 5 gives representative blanket performance for three fuels and various exposures. The last three columns of this table lists effective time average performance for blankets with three fuel types after various exposures. The first five columns list local instantaneous blanket values. Directing your attention to the UC fuel five-year exposure case, we see that as the local T breeding ratio increases from 1.05 to 1.43, the average blanket tritium breeding ratio is 1.05. Local M increases from 8 to 13.6 giving an average M of 9.18. Local net plutonium breeding decreases from 1.38 to 1.18 giving an average net plutonium breeding ratio of 1.09. At five-years exposure, burnup is 1.5% and plutonium 239 buildup is 2.9%. As exposure continues effective T breeding and energy multiplication increase while effective fissile breeding decreases.

The U-moly case is representative of metallic fuels such as uranium silicide. The thorium case gives the lowest blanket performance, but the higher conversion ratio of uranium 233 fuel in thermal fission reactors makes thorium competitive with metallic uranium fuel blankets. Combining the advantages of uranium and thorium in a single blanket should give the best performance.

Fissile accumulation in these three blankets are graphically displayed in Fig. 11. After an exposure of five-megawatt years per square meter, one square meter of blanket accumulates 50 kilograms of plutonium for the U-moly case, 35 kilograms of plutonium for the uranium carbide case and 20 kilograms of uranium 233 for the thorium case.

3. FUSION DRIVER:

The central element of any fusion-fission system is the fusion neutron source. Fusion physics research programs have been underway since the early 50's. This effort is international in scope and has as its goal commercial fusion power.

The principle approaches to fusion being pursued are the Tokamak and mirror magnetic confinement fusion, and laser and electron beam heated inertial confinement fusion. Both magnetic and inertial confinement fusion are being studied as possible neutron sources for fusion-fission. While I will limit my specific remarks to magnetic mirror systems, the energetics of mirror based fusion-fission is representative of fusion-fission systems in general.

In all fusion systems fuel must be heated to fusion temperatures and confined at a high enough product of density and time ($n\tau$) for significant fusion to occur. We call the ratio of fusion energy produced to energy invested in the plasma, Q.

The conceptual basis for magnetic mirror confinement fusion derives from the fact that energetic charged particles can be trapped on the field lines of a properly shaped magnetic field, provided the field is intense enough. In *mirror machines*, the field lines enter and leave the confinement zone through the chamber ends. Particle escape along the field lines is inhibited by intensifying the end fields. Particles approaching these regions (the *mirrors*) are then reflected, provided they are not directed too nearly parallel to the field lines. Escape through the mirrors occurs when interparticle collisions or other processes deflect a particle so that its vector is nearly parallel to the field lines.

The coil and plasma geometry needed for MHD (magnetohydrodynamic) stable mirror confinement is displayed in Fig. 12. The magnetic field in this geometry increases in all directions outward from the center thus is called minimum-B mirror geometry.

Progress in mirror plasma physics experiments has been rapid within the last few years. The next generation mirror experiment MFTF scheduled for operation in 1981 is expected to achieve physics parameters close to a reactor-grade plasma. In Table 6 we list key physics parameters for the present mirror experiment, 2XIIB, for the next experiment, MFTF, and for a commercial hybrid reactor. The 2XIIB experiment has 20 and 40 keV neutral beam injectors; MFTF will have 20 keV injectors for start-up and 80 keV injectors for sustained (1/2 sec) operation. The major difference between the MFTF plasma and that required for a commercial reactor is physical size and duty factor; MFTF however is a sufficiently versatile experiment that it should establish scaling laws that will allow us to extrapolate to the larger plasma with a high degree of confidence. Also, $n\tau$ for the commercial facility is about a factor of 10 larger than in MFTF, this being a result of higher energy and the D-T isotopic mixture used in the commercial reactor as compared to MFTF. Thus, achievement of the anticipated plasma performance from MFTF will provide an adequate physics base from which development of a mirror hybrid could proceed.

4. REVIEW OF MIRROR HYBRID STUDIES:

Our work on the mirror hybrid concept started in the early 70's. This work started by examining the nuclear performance of blankets containing fertile materials.[1,5,6] During this early stage we performed general scoping studies[7] and completed our first reactor study. This study concentrated on a high M thermal fission blanket.[8]

In 1975 we performed our first conceptual reactor design study.[9] This design had a fast fission uranium carbide blanket. Following completion of this conceptual design, we concentrated on optimization of mirror hybrid fusion-fission systems.[10,11,12,13] Results of these studies helped to convince us that fissile fuel production would be an attractive role for mirror hybrid reactors.

In 1977 we and an industrial partner (General Atomic Company) joined forces to perform a more detailed point design study of a Pu 239 producing standard mirror hybrid reactor.[14]

5. SYSTEM PERFORMANCE:

The economic performance of a fusion-fission power system depends on the type and performance of the fusion driver, blanket and fission reactors that comprise the system. As an example, I will outline our estimate of the performance of two systems; one based on the U/Pu fuel cycle, the other on the Th/U fuel cycle. Both use the standard minimum-B mirror as the fusion driver. The U/Pu based system is a uranium blanket producing makeup Pu 239 for light water reactors (LWR's). The Th/U based system uses a thorium blanket producing makeup U-233 for high temperature gas reactors (HTGR's). The blankets are the fast fission hybrid blankets (U-Mo), and Th) described in the blanket section (Table 6).

Both hybrid reactors are optimized to minimize the cost of electricity from the fusion-fission system. Both are sized to have thermal power ratings approximately the same as present day fission reactors.

The performance parameters for both systems are summarized in the following six tables (Table 7-12).[illegible]

The optimized reactor parameters with uranium and thorium blankets are listed in Table 7. There are several significant differences between the two reactors. The uranium blanket, because of its high energy multiplication, results in a plant with a large electrical output. The thorium blanket reactor does not produce net electricity, just fissile fuel. Both blankets have about the same thermal rating, this being the result of a much larger fusion power output from the thorium blanket reactor as compared to the uranium blanket reactor. The high fusion power of the thorium blanket reactor is obtained by using a more intense magnetic field than for the uranium blanket reactor. The uranium blanket reactor may, therefore, rely on existing NbTi superconductor magnet technology, whereas the thorium blanket reactor will require the more technologically advanced Nb_3Sn superconductor.

The blanket parameters for the optimized reactors are listed in Table 8. Both produce between 2 and 3 metric tons of fissile fuel per year. However, the thorium blanket requires a rather high exposure, and the possibility of the blanket structure being able to attain ~ 9 MW-yr/m^2 exposure is more uncertain. The average energy multiplication of the uranium blanket is about a factor of 4 higher than for the thorium blanket; these blanket multiplications include the effect of the fractional blanket coverage.

The fission reactors chosen as burners of the hybrid fissile fuel are listed in Table 9, along with their requirements for hybrid fissile fuel. As a burner of Pu, we have used a light water reactor (LWR) on a Pu recycle fuel cycle and supplemented with hybrid Pu. As a ^{233}U burner we have used a high-gain HTGR, using the thorium -^{233}U fuel cycle.

The hybrid economic parameters are listed in Table 10. The higher capital cost of the thorium blanket hybrid is associated with the fusion components required to generate the higher fusion power than the uranium blanket hybrid. The ^{233}U cost is more than a factor of two greater than the Pu value. However, the lower fissile requirements of the HTGR as compared to the LWR results in approximately the same electricity value from the two fission power plants. The breakdown of the fissile material costs indicate that they are dominated by capital costs. The fuel cycle costs account for fabrication, reprocessing and spent fuel shipping. Current (high) estimates for these services have been used, but they are not a dominate cost. For the uranium blanket reactor, approximately 60% of the plant revenues are generated by fissile production in contrast to the total revenue generation by fissile material for the thorium hybrid.

The fission reactor economics are listed in Table 11. The important result here is that the cost of producing fissile fuel in the hybrid (4.0 and 5.3 mills/kw-hr) is a small fraction of the total electricity cost. The conclusion is that the mirror hybrid reactor, based on minimum-B mirror plasma confinement, is capable of converting the large fertile resources of the world into fissile fuel at a cost that does not strongly influence the net cost of electricity. A factor of two increase in fissile fuel generation costs (i.e., cost of the hybrid) would only increase electricity cost by about 20%.

Installed capacity, capital cost and electric costs of both fusion-fission systems are summarized in Table 12. The uranium blanket hybrid supports 17 Mwe per Mw fusion while the thorium hybrid supports 9.

6. REFERENCE DESIGN:

Based on the results of our optimization studies and on successes in the mirror physics program we spent 1977 developing a reference design of a mirror (fusion-fission) hybrid reactor.[7] Our objectives were to conceptually design an early commercial facility for Pu 239 generation based on near-term physics and technology.

To summarize the reference mirror hybrid design, we list the major design choices that have been made for the reactor.

- Minimum-B mirror confinement
- Yin-Yang coil design, NbTi superconductor
- positive ion injectors with direct recovery
- fast spectrum blanket neutronics
- single-stage plasma direct converter
- cryocondensation vacuum pumping
- Blanket
  - U_3Si fuel (depleted U)
  - LiH tritium breeder (natural Li)
  - Inconel 718 structural material
- He primary heat transfer loop (PHTL)
- Prestressed Concrete Reactor Vessel (PCRV)

magnet restraint

PHTL restraint

blanket support and restraint

. steam thermal conversion system

Characteristics for the reactor are listed in Table 13.

The plasma modeling effort for the hybrid this year is using the full range of analytical tools that have been developed in support of mirror physics investigations in the 2XIIB experiment.[15] These tools include the MCFLS equilibrium code,[16] a two-dimensional (in velocity space) Fokker-Planck code,[17] and a radial transport code.[18,19]

The variation of the basic Yin-Yang magnet, developed for reactor applications, is shown in Fig. 13. This design uses a large main coil with a small mirror coil inside and has the virtue of locating the mirror point very near the inside surface of the conductor. This location for the mirror point implies a minimum size opening in the blanket for the plasma leakage fans. The magnet has an outside diameter of about 22 meters, and a distance of 13 meters between mirror points. It is designed with a maximum field at the conductor of 8 Telsa, dictated by the use of NbTi superconductor. The maximum current density is about 10^3 A/cm^2. These conditions imply comparatively modest magnet technology, although the magnet is quite large, about 5500 tonnes for each magnet half (including the stainless steel coil case).

The neutral beam injector design developed for the reference hybrid is based on the positive ion injector developed at the Lawrence Livermore Laboratories (LBL-LLL) with the following modifications. A hollow cathode ion source is proposed as a means of providing a much longer-lived cathode than the heated tungsten wire cathode now used in the LBL-LLL source. A mechanical design for the sources has been developed which will place all high-voltage insulators out of the path of the direct 14 MeV neutron flux. A Hg vapor jet at the end of the neutralizer region strongly impedes gas flow from the source into the downstream regions of the beam line, resulting in much lower background pressure in the beam line and reduced cryopumping requirements. High pressure vacuum pumping in the region of the sources will use Hg diffusion pumps. A cryopanel design has been developed which will permit continuous operation of the cryopumping equipment in the beam lines. The design permits defrosting of some of the cryopanels while the remainder are still in operation.

The reactor requires deuterium injectors with acceleration to 115 keV and tritium to 187 keV. When account is taken of the half and third energy components in the beam, the average beam energies are 94 and 141 keV, respectively, for D^{0} and T^{0}. Our analysis predicts an efficiency for the injectors of 65%.

Outboard of the coils, end tanks must be provided to receive the plasma leakage. In the end tanks, we perform direct conversion, converting some of the kinetic energy of the ion flow directly into electricity. The remaining kinetic energy is deposited as thermal energy in the direct converter electrodes and must be removed by active cooling. Upon striking the direct converter electrodes, the plasma flow is neutralized and the end tank contains vacuum pumping equipment to remove the resulting gas load.

To provide access to the blanket from outside the machine, it is a convenient design feature to have one of the end tanks as small as possible. We implement the small end tank by designing the magnet such that one of the mirror fields is 5% stronger than the other. This field perturbation causes approximately 90% of the plasma leakage to flow out through the weak mirror and the remaining 10% to exit through the strong mirror. Since the size of the end tank is proportional to the amount of plasma flow, we can use a small end tank on the strong mirror. To keep this tank as simple as possible, we do not perform any direct conversion but design for the plasma energy to be deposited as thermal energy, with provisions for active cooling and vacuum pumping with cryopanels. The large end tank, which receives the 90% end leakage flow, is designed with a simple single-stage direct converter unit, having an upper limit on its efficiency of about 50%. This end tank must also have provisions for active cooling and vacuum pumping.

In the past, we have examined the use of primarily three fertile fuels in the blanket: UC, U-Mo alloy, and thorium. In our present hybrid design we are advocating the use of U_3Si, a fuel being developed in the Canadian nuclear power program for the CANDU reactor. Our reasons for this choice are: 1) high uranium density (U_3Si is a metallic alloy), 2) ease of fabrication, and 3) a comparatively high burnup capability (for a metallic fuel) on the order of 2-3%. Economic optimization of the fuel cycle for this reactor to attain a total fuel exposure of about 15 MW-yr/m^2 of 14 MeV neutron energy through the first wall. In Table 14, the initial (beginning of life) and final (end of life) neutronic parameters for the UO$_2$ blanket are listed.

One of our primary concerns in the mechanical design of the reactor was to provide highly reliable support and containment of the blanket and primary heat transfer loop components. The basis of our concern was the conclusion that the primary safety consideration for the reactor was a loss of flow accident, and the design therefore had to be one in which the maintenance of forced cooling to the blanket could be assured to a high level of confidence.

The design approach we have selected is to mount the magnet, blanket and primary heat transfer loop all within a prestressed concrete reactor vessel (PCRV) of the type developed for gas-cooled fission reactors. This is shown in Fig. 14. In the center of the PCRV is the magnet and blanket, and the steam generators and He circulators are located around the periphery. The blanket is a spherical shell inside the magnet. In this way, the blanket and its cooling system are locked together so that no relative motion between them can occur, thus precluding the possibility of rupturing any of the coolant ducts.

The PCRV also serves a second function. It provides the main restraining forces for the magnet. Since the PCRV operates at room temperature, a way had to be found to transmit the forces from the magnet (at 4°K) out to the concrete. Our design solution has been to use a high-compressive-strength thermal insulation (a silicate refractory) capable of sustaining about 5,000 psi. Our calculations have shown that an insulation thickness of about 50 cm is adequate to reduce the heat leak from the concrete to the magnet to an acceptable value.

The blanket design concept is one which avoids any major disassembly of the reactor during the blanket change operation but instead relies on remote operations to assemble and disassemble the blanket inside the PCRV. The blanket is made up of small cylindrical modules, approximately 50 cm in diameter, with the blanket structure being suspended directly from the inside wall of the PCRV as shown in Fig. 15. Removal and replacement of blanket modules is accomplished with the refueling machine shown in Fig. 16, which consists of a post which is inserted down through the center of the machine and has a pivoting arm to operate on the modules. The blanket replacement operation consists of a series of manipulations of each of the several hundred modules.

The module, as shown in Fig. 17, consists of a cylindrical pressure vessel with a hexagonal base. One of the more challenging aspects of the module design has been to devise a fast, reliable method of making up the seal that isolates the high pressure He coolant from the vacuum region that contains the plasma. We have discarded a welded joint, since remote grinding and welding are time consuming operations and we have serious doubts about the ability to consistently generate remote vacuum-tight welds. We have therefore adopted a bolted joint using a double knife-edge seal with differential pumping between the two knife-edges. The pressure vessel is bolted in place with 6 bolts, one at each corner of the hex-shaped base. The internals of the module are fabricated as a single unit, containing the U-Pu pins, the tritium breeding unit and internal flow ducting. Thus, to renew a module the pressure vessel is unbolted and removed, the pin assembly is removed, a new pin assembly is inserted and a new pressure vessel is bolted in place. The coolant flow is re-entrant, with the [illegible] U-Pu pins being cooled by the inlet flow and the coolant then proceeding down to the first wall, turning, and cooling the [illegible] U-Pu pins on its exit path out through the module.

The primary heat transfer loop is designed to operate with helium as the working fluid. The coolant pressure is 50 atm., with an inlet temperature to the blanket of 300°C and an outlet temperature of 600°C. The flow path is designed to maintain the relative pressure drop, Δp/p, to about 3% through the entire loop (blanket, ducting and steam generator). This combination of conditions permits the use of existing gas-cooled fission reactor technology for the design of the He circulators and steam generators.

The local blanket multiplication and therefore local blanket power density increases by about a factor of two over the life of the fuel. By devising an appropriate fuel management scheme for the blanket, we are able to limit the peak-to-average variation in the total blanket thermal power to about 10% (3,600 MW average; 4,000 MW peak) and the primary heat transfer and power conversion loop capacity are designed to accommodate this power variation. The blanket modules are grouped into four quadrants and at time intervals of one quarter of the blanket life, the reactor is shut down and one quadrant of the blanket is refurbished with new fuel assemblies. In this way we are able to establish an equilibrium fuel cycle where the four quadrants are each at a different exposure.

The thermal-hydraulic design for the fuel, on the other hand, must provide adequate cooling of the fuel pins during the lifetime power density variation of [illegible]. Our present design specifies a peak fuel power density level, at the first wall (at beginning-of-life) of about [illegible] watts cm^{-3} and an end-of-life value of [illegible] watts cm^{-3}. The fuel pins are [illegible] cm in diameter with [illegible] mm thick Inconel 718 clad in a pitch-to-diameter ratio of [illegible]. The maximum inner wall clad temperature (hot channel) is limited to [illegible]°C.

In this section we have summarized the reference mirror hybrid reactor design performed by LLL and General Atomic Company. The reactor parameters have been chosen to minimize the cost of producing fissile fuel for consumption in fission power reactors. As in the past, we have emphasized the use of existing technology where possible and a minimum extrapolation of technology otherwise. The resulting reactor may thus be viewed as a comparatively near-term goal of the fusion program, and we project improved performance for the hybrid in the future as more advanced technology becomes available.

7. FUTURE STUDIES

For future mirror hybrid studies we are planning to adopt new mirror confinement concepts and concentrate on U-233 producing blankets. One such new confinement concept is the Tandem Mirror. Our predictions are that a low technology ([illegible] T magnets and ~[illegible] keV injectors) tandem might have a Q near [illegible] and a wall loading near [illegible] MW m^{-2}. Our past studies have indicated such Q's and wall loading should make an excellent fusion driver for a hybrid reactor. A tandem mirror experiment (TMX) is under construction at LLL to test the basic physics of the tandem concept. This experiment should be operating by the end of [illegible].

We will concentrate on U-233 producing blankets because of the importance placed on proliferation resistant fuel cycles. We believe such fuel cycles will only be practical if there is an external source of fissile material to supplement the poor neutron economy expected of proliferation resistant fuel cycles.

[illegible]

[illegible]

[illegible]

[illegible]

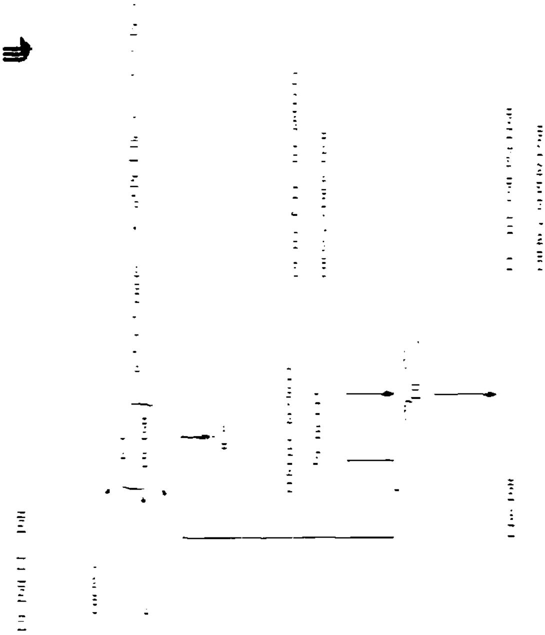
[illegible]

[illegible]

[illegible]

[illegible]

[illegible]



FUSION-FISSION SYSTEMS

SYMBIOSIS		HYBRID
D + T FUSION	FUSION COMPONENTS	D + T FUSION
N (14 MeV)		N (14 MeV)
	BLANKET	
^{7}Li (N,N' α) T		^{238}U (N,FISS) (N,2N) (N,3N)
AND/OR ^{9}BE(N,2N)		AND/OR ^{232}TH(N,FISS) (N,2N) (N,3N)
NEUTRONS		NEUTRONS
^{6}Li (N, α) T		^{6}Li (N, α) T
		^{238}U (N, γ) AND/OR
^{232}TH (N, γ) ^{233}U		^{232}TH (N, γ)
^{233}U		^{239}PU AND/OR ^{233}U
		^{239}PU AND/OR ^{232}U (N,FISS)
^{232}U	SEPARATE FISSION REACTORS	^{239}PU AND/OR ^{233}U
(N,FISS.)		(N,FISS.)

Figure 2

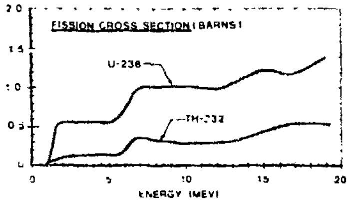


Figure 4

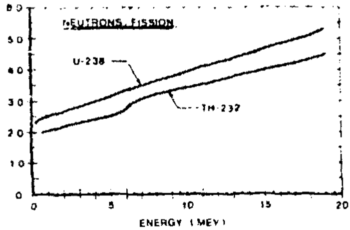


Figure 5

BLANKET COVERAGE

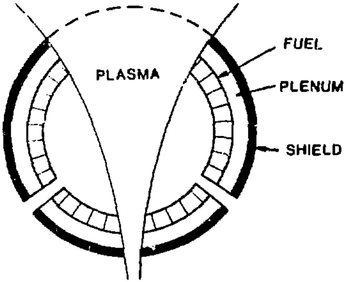


Figure 8

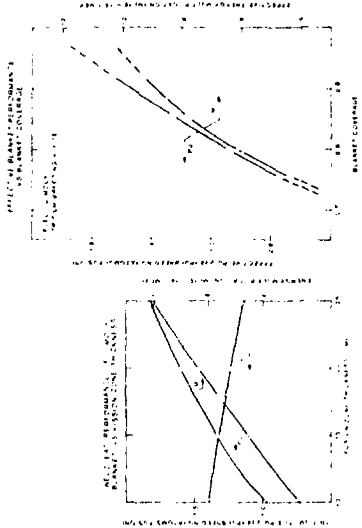


Figure 9

UC BLANKET PERFORMANCE VS. EXPOSURE

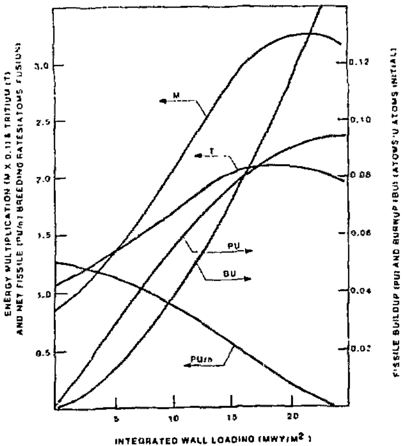


Figure 10

FISSILE (PU 239 OR U233) ACCUMULATION VS EXPOSURE

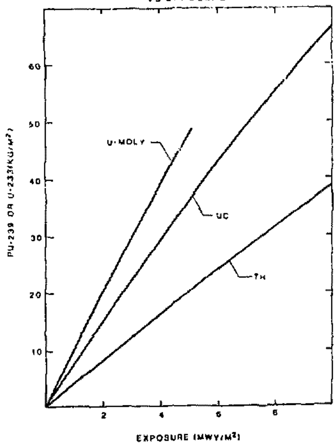


Figure 11

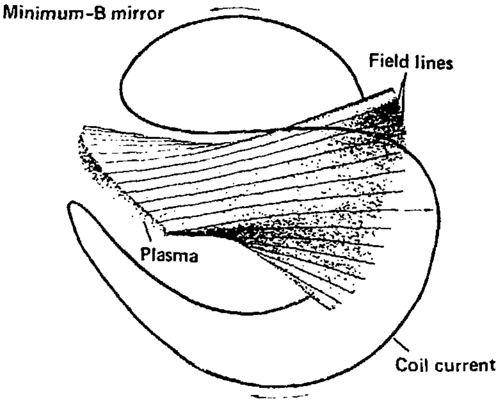


Figure 12
MINIMUM-B MIRROR
GEOMETRY

MAGNET CONDUCTOR CONFIGURATION

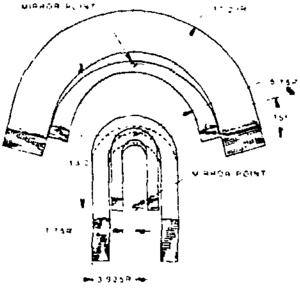


Figure 13

FUSION-FISSION MIRROR HYBRID REACTOR

LAWRENCE LIVERMORE LABORATORY

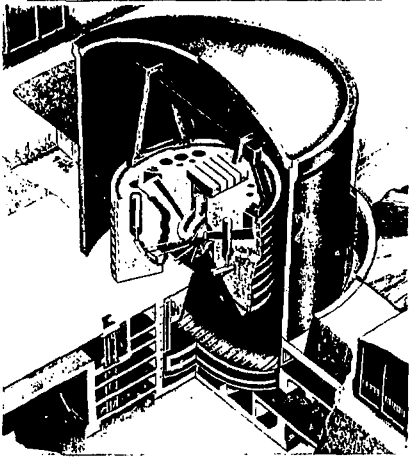


Figure 14

BLANKET/SHIELD CUTAWAY HYBRID REACTOR

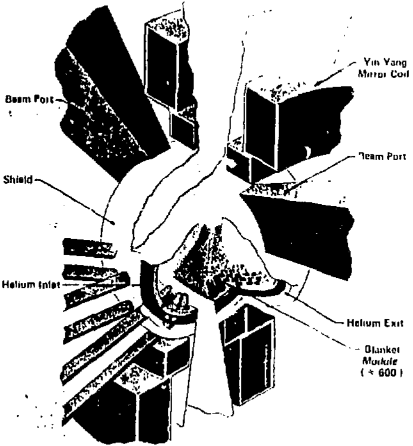
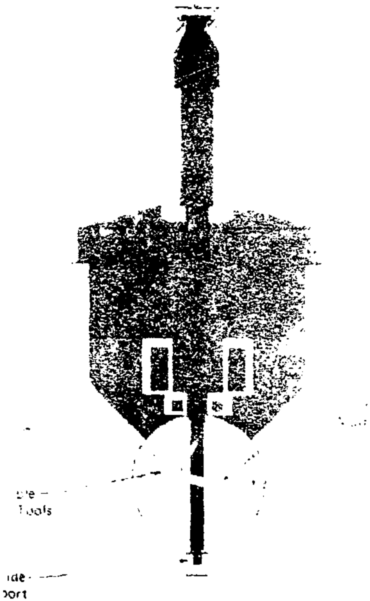
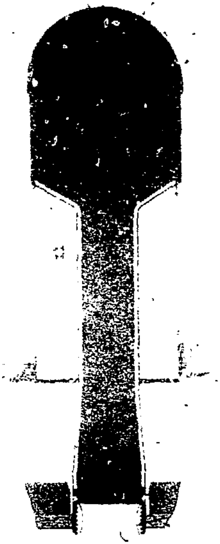


Figure 15





REPRESENTATIVE FAST FISSION BLANKET PERFORMANCE

	LOCAL BLANKET VALUES					BLANKET AT EQUILIBRIUM[4]		
FUEL & EXPOSURE[1]	T/N	M	Pu/N	(BURN UP(%)[2]	BUILDUP(%)[3]	T/N	M	Pu/N
UC AT 0 YEARS	1.05	.01	1.33	0	0	-	-	-
AT 5 YEARS	1.45	13.6	1.18	1.5	2.9	1.05	9.13	1.09
AT 10 YEARS	1.76	21.6	0.66	3.7	5.5	1.20	12.6	0.96
U-MOLY AT 0 YEARS	1.10	10.0	1.80	0	[illegible]	-	-	-
AT 5 YEARS	1.60	20.0	1.70 (U/n)	1.3	2.5	1.15	12.7	1.49 (U/n)
TH AT 0 YEARS	1.08	2.5	0.77	0	0	-	-	-
AT 12 YEARS	1.23	5.4	0.61	0.74	2.76	0.98	3.4	0.57

1 EXPOSURE AT A FIRST WALL OF NEUTRON ENERGY FLUX OF 1.0 MW/M^2

2 FISSION OF HEAVY METAL

3 NET ACCUMULATION OF FISSILE MATERIAL IN (%) OF INITIAL HEAVY METAL

4 TIME AVERAGED PERFORMANCE OF BLANKET WITH 85% COVERAGE

TABLE 5

PLASMA PARAMETERS

Table 6

	2XII-B	MFTF	COMM. HYBRID
Plasma Length (M)	1.6	3.4	13
Conductor Field (T)	-	7	5
β	0.44 0.7	0.5	0.7
Injection Energy (keV)	20.0 (40.0)	80.0 (20.0)	125.0
$n\tau$ (s/cm^3)	7×10^{10}	10^{12}	10^{13}
L/a_i	35.0	2.0-200	[illegible]
R_p/a_i	3.0	10-50	50.0

Table 2. Parameters for the optimized hybrid reactors.

Parameter	U/Mo	Th
Mirror ratio	2.50	2.75
Injection energy (keV)	100	100
Conductor field (T)	8	12
Q	0.68	0.75
Fusion power (MW)	470	1500
First wall flux (MW/m^2)	1.3	4.2
Blanket thermal power,		
average (MW)	4220	3340
Electrical output (MW)	1040	-40
Capacity factor	0.75	0.73
Mirror-to-mirror length (m)	15	15

Table 3. Blanket parameters for the optimized reactors.

Parameter	U/Mo	Th
Fissile output (kg/yr)	2363	2590
Avg. energy multiplication	11.1	2.3
Blanket coverage	0.86	0.77
Fertile burnup (%)	1.9	0.5
Blanket exposure (MW-yr/m^2)	4.1	9.2
Fuel power density (W/cm^3)	150	110
Blanket enrichment, avg.	1.02%	1.08%

Table 9. Description of thermal converter reactors

Parameter	Burner ^{239}Pu	Burner ^{233}U
Reactor type	LWR	High-gain HTGR
Fuel cycle		
Fertile feed	Natural U	Th
Fissile feed	Pu	^{233}U
Fissile recycle	Pu	^{233}U
Conversion ratio	0.6	0.8
Fissile feed requirement (kg/yr/MWe)	0.333	0.135

Table 10. Economics for the optimized hybrid reactors

Cost	U/Mo	Th
Capital cost (10^9 \$)	2.3	3.3
(\$/kWe)	2200	-
Fissile mat. cost (\$/g)	55	127
Capital	80	103
Fuel cycle	13	21
Operation & Maintenance	1	1
Electricity revenues	-39	2
Electricity cost (mills/kW-hr)	24.8	25.3

Table 11. Economics for the fission reactors.

Cost	LWR	High-gain HTGR
Capital cost (\$/kWe)	750	750
Electricity cost (mills/kW-hr)	24.8	25.3
Capital cost	16.1	16.1
Fuel cycle without fissile material	3.9	3.2
Fissile fuel	4.1	5.3
Operation & Maintenance	0.7	0.7

Table 12. Economics for the hybrid/thermal reactor system.

Cost	U/Mo	Th
Installed capacity (MWe)	8130	14 000
Hybrid	1040	-
Fission reactors	7090	14 000
Capital cost (\$/kWe)	935	985
Electricity cost (mills/kW-hr)	24.8	25.3
Capital	19.7	20.5
Fuel cycle	4.3	4.0
Operation & mainten.	0.8	0.8

REACTOR CHARACTERISTICS

Table 13

Fusion Power	400 MW
Thermal Power (Avg.)	3600 MW
Injected Neutral Power	625 MW
Net Electric Output Power	525 MW
First Wall 14 MeV Neutron Current	2 MW/m^2
Fissile Production Rate	2700 kg/yr
Injection Energy D^0	125 keV
T^0	187 keV
β	0.7
Central Ion Density	9×10^{13} cm^{-3}
Q	0.63
$n\tau$	2×10^{13} sec/cm^3

TIME-DEPENDENT U$_3$Si BLANKET NEUTRONIC PARAMETERS

Table 14

Exposure ($MW\text{-}yr/m^2$)	M	Pu/n	% Pu	Burnup %	T/n
0	8.8	1.65	0	0	1.05
5	19.4	1.75	2.3	0.75	1.42